

Virginia Electric and Power Company  
North Anna Power Station  
P. O. Box 402  
Mineral, Virginia 23117

December 23, 1999

U. S. Nuclear Regulatory Commission  
Attention: Document Control Desk  
Washington, D. C. 20555

Serial No.: 99-607  
NAPS: JHL  
Docket No.: 50-339  
License No.: NPF-7


Dear Sirs:

Pursuant to 10CFR50.73, Virginia Electric and Power Company hereby submits the following Licensee Event Report applicable to North Anna Unit 2.

Report No. 50-339/99-004-00

This report has been reviewed by the Station Nuclear Safety and Operating Committee and will be forwarded to the Management Safety Review Committee for its review.

Very truly yours,

  
W. R. Matthews  
Site Vice President

Commitments contained in this letter: None

Enclosure

cc: U. S. Nuclear Regulatory Commission  
Region II  
Atlanta Federal Center  
61 Forsyth Street, SW, Suite 23T85  
Atlanta, Georgia 30303  
  
Mr. M. J. Morgan  
NRC Senior Resident Inspector  
North Anna Power Station

IE22

PD2 ADDU 05000339

## LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

Estimated burden per response to comply with this mandatory information collection request, 50 hrs. Reported lessons learned are incorporated into the licensing process and fed back to industry. Forward comments regarding burden estimate to the Records Management Branch (T-6 P33), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, and to the Paperwork Reduction Project (3150-0104), Office of Management and Budget, Washington, DC 20503. If an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

FACILITY NAME (1) <b>NORTH ANNA POWER STATION, UNIT 2</b>										DOCKET NUMBER (2) <b>05000339</b>		PAGE (3) <b>1 OF 4</b>		
TITLE (4) <b>MANUAL REACTOR TRIP DUE TO LOSS OF FEEDWATER PUMP SUCTION PRESSURE AND AUXILIARY FEEDWATER ACTUATION</b>														
EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)					
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME			DOCUMENT NUMBER		
12	02	1999	1999	004	00	12	23	1999	FACILITY NAME			DOCUMENT NUMBER		
												05000-		
												05000-		
OPERATING MODE (9)		THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR 5: (Check one or more) (11)												
1		20.2201(b)			20.2203(a)(2)(v)			50.73(a)(2)(i)			50.73(a)(2)(vii)			
POWER LEVEL (10)		98 %			20.2203(a)(1)			20.2203(a)(3)(i)			50.73(a)(2)(ii)			
					20.2203(a)(2)(i)			20.2203(a)(3)(ii)			50.73(a)(2)(iii)			
					20.2203(a)(2)(ii)			20.2203(a)(4)			x 50.73(a)(2)(iv)			
					20.2203(a)(2)(iii)			50.36(c)(1)			50.73(a)(2)(v)			
					20.2203(a)(2)(iv)			50.36(c)(2)			50.73(a)(2)(vii)			
											Specify in Abstract below or in NRC Form 366A			
LICENSEE CONTACT FOR THIS LER (12)														
NAME <b>W. R. Matthews, Site Vice President</b>										TELEPHONE NUMBER (Include Area Code) <b>(540) 894-2101</b>				
COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)														
CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX		CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX				
SUPPLEMENTAL REPORT EXPECTED (14)										EXPECTED SUBMISSION DATE		MONTH	DAY	YEAR
YES (If yes, complete EXPECTED SUBMISSION DATE)										X NO				
ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)														
<p>On December 2, 1999, at 1647 hours, with Unit 2 in Mode 1 operating at approximately 98% power, the reactor was manually tripped due to the loss of two of three main feedwater pumps. The main feedwater pumps tripped on low suction pressure due to the loss of discharge flow from the "A" high pressure heater drain pump following a secondary plant transient. The auxiliary feedwater pumps automatically actuated following the reactor trip to restore inventory in the steam generators. This event is reportable pursuant to 10CFR50.73 (a)(2)(iv) for any event or condition that resulted in a manual or automatic actuation of any engineered safety feature (ESF), including the reactor protection system (RPS).</p> <p>The cause of the event is attributable to the failure of a feedwater heater level control switch to properly actuate. A root cause evaluation of the event is being performed.</p> <p>This event posed no significant safety implications. The reactor was safely shut down and ESF equipment operated properly. The health and safety of the public were not affected.</p>														

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TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

**1.0 DESCRIPTION OF THE EVENT**

On December 2, 1999, at 1647 hours, with Unit 2 in Mode 1 operating at approximately 98% power, the reactor was manually tripped due to the loss of the "A" and "B" main feedwater pumps (2-FW-P-1A and B) (EIS System - SJ, Component - P). The "C" main feedwater pump (2-FW-P-1C) remained in operation during the event.

The initiation of the event (occurring with Unit 2 at 100% power) was the closure of the "1A" feedwater heat exchanger (2-FW-E-1A) (EIS Component - HX) cascade valve (2-SD-LCV-203A) (EIS System - SN, Component - LCV) to the "2A" feedwater heat exchanger (2-FW-E-2A). This resulted in a 2-FW-E-1A Hi/Hi level alarm and a reduction in main feedwater pump suction pressure. The third condensate pump (2-CN-P-1C) (EIS System - SD) was previously tagged, due to concern for a potential failure of the suction expansion joint (EIS Component - EXJ) which was visibly deformed, and was unavailable to completely restore feedwater suction pressure. The Hi/Hi level in the 2-FW-E-1A resulted in the closure of the first point extraction steam motor-operated valve (2-ES-MOV-201A) (EIS System - SE, Component - V) causing cooler water to enter the steam generators (EIS System - AB). The highest power level indicated by nuclear instrumentation was 101.2%. A ramp down to approximately 98% power was implemented. At this point, the "1A" feedwater heat exchanger cascade valve (2-SD-LCV-203A) was manually isolated.

While investigating the closure of 2-SD-LCV-203A, a 2-FW-E-2A Hi/Hi level alarm was received when the level switch cover was removed. The second point extraction steam motor-operated valve (2-ES-MOV-202A) isolated due to the Hi/Hi level alarm in the "2A" feedwater heat exchanger. A ramp down of 1% per minute was initiated. Approximately one minute later, 2-FW-P-1A tripped on low suction pressure and the "1B" main feedwater pump (2-FW-P-1B) automatically started. A 4% percent per minute ramp down was initiated. The "A" high pressure heater drain pump (2-SD-P-1A) (EIS System - SN), which pumps heater drains into the feed pump suction line, was then manually stopped due to low and fluctuating discharge pressure. Main feedwater pump 2-FW-P-1B subsequently tripped on low suction pressure. At 1647 hours, the reactor was manually tripped due to the loss of two of three main feedwater pumps on low suction pressure. The auxiliary feedwater pumps automatically started following the manual reactor trip and were secured upon restoration of steam generator level. The "C" main feedwater pump remained in service during the event to provide feed flow to the steam generators. The steam dumps opened to the condenser (EIS Component - COND) to remove excess heat.

Control Room personnel responded to the reactor trip in accordance with procedure 2-E-0, Reactor Trip or Safety Injection. Primary plant response was normal for this transient. Initially, Reactor Coolant System (RCS) pressure decreased to approximately 1970 psig.



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TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

pressurizer level decreased to 24%, and RCS temperature decreased to approximately 546 degrees F. Pressurizer pressure, level and RCS temperature subsequently returned to their normal programmed values. All ESF equipment responded as designed.

On December 2, 1999, at 1811 hours, a four hour report was provided to the NRC per 10 CFR 50.72 (b)(2)(ii) due to the reactor trip and auxiliary feedwater pump actuation. This event is reportable pursuant to 10CFR50.73 (a)(2)(iv) for any event or condition that resulted in a manual or automatic actuation of any engineered safety feature (ESF), including the reactor protection system (RPS).

**2.0 SIGNIFICANT SAFETY CONSEQUENCES AND IMPLICATIONS**

This event posed no significant safety implications. The reactor was safely shut down and ESF equipment operated properly. Therefore, the health and safety of the public were not affected at any time during this event. The ability to remove heat by normal secondary plant equipment remained in service during this event.

**3.0 CAUSE**

The cause of the event is attributable to the failure of a feedwater heater level control switch to properly actuate. A root cause evaluation of the event is being performed.

**4.0 IMMEDIATE CORRECTIVE ACTION(S)**

Following the reactor trip, Operations Emergency Procedure 2-E-0, Reactor Trip or Safety Injection was entered and immediate actions were performed to bring Unit 2 to a safe stable condition. The post trip response progressed as expected and the operators transitioned to 2-ES-0.1, Post Trip Recovery. Unit 2 was stabilized at no-load conditions.

**5.0 ADDITIONAL CORRECTIVE ACTIONS**

A "Post Trip Review" meeting was conducted with station personnel, on December 2, 1999, to identify the cause of the event, identify abnormal or degraded indications occurring during the reactor trip, and to assess Unit readiness for return to operation.

Equipment malfunctions were appropriately dispositioned and Unit 2 was returned to power operations on December 3, 1999. A root cause evaluation of the event is being performed. Following completion of the root cause evaluation, any additional recommended corrective actions will be evaluated and implemented, as required.

**6.0 ACTIONS TO PREVENT RECURRENCE**

None.

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TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

**7.0 SIMILAR EVENTS**

None.

**8.0 ADDITIONAL INFORMATION**

Unit 1 was at 100% power and not affected by this event.